NON-PUBLIC?: N
ACCESSION #: 8901190387
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Rancho Seco Nuclear Generating Station PAGE: 1 OF 8

DOCKET NUMBER: 05000312

TITLE: Manual Reactor Trip Due To Two Pressure Control Valve Failures
EVENT DATE: 12/12/88 LER #: 88-019-00 REPORT DATE: 01/11/89

OPERATING MODE: N POWER LEVEL: 012

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: Steve Rutter, Supervisor, Independent Investigation/Reviews

TELEPHONE: (916) 452-3211

COMPONENT FAILURE DESCRIPTION:
CAUSE: X SYSTEM: SA COMPONENT: PCV MANUFACTURER: F130
X SA PCV F130
B SB ISV P032
REPORTABLE TO NPRDS: N
N
N

SUPPLEMENTAL REPORT EXPECTED: NO EXPECTED SUBMISSION DATE:

ABSTRACT:

At approximately 1433 hours on December 12, 1988, Control Room operators manually tripped the reactor from 12% power. Operations personnel were manually controlling auxiliary steam header pressure because two pressure control valves had failed 4 hours earlier. During manual control, a loss of steam pressure to the 'B' main feedwater pump caused the loss of feedwater flow and subsequently caused the Once-Through Steam Generator (OTSG) level to drop to the Emergency Feedwater Initiation and Control (EFIC) low level initiation setpoint for 'B' auxiliary feedwater (AFW). Having observed the initiation of EFIC, Control Room operators followed Casualty Procedure C.10 "Loss of Steam Generator Feed" and tripped the plant. Operators observed the reactor cooldown outside the post-trip window and isolated the AFW supply to the 'B' OTSG in accordance with emergency procedures. This resulted in the

'B' OTSG boiling dry, terminating the reactor cooldown.

At approximately 1500 hours, the operators restored AFW flow to the OTSG at a rate of 50 gpm. By 1530 hours, the 'B' OTSG pressure had returned to normal.

There was no damage to plant equipment as a result of the valve failures involved and subsequent plant trip. There was no impact to the health or safety of the public due to this event.

END OF ABSTRACT

Description of the Event

(Refer to Figure 1 for a diagram of the auxiliary steam path to the main feedwater turbines.)

At 0024 hours on December 12, 1988, Control Room operators took the reactor critical following a 3 day outage. At 0440 hours, the pressure reducing station from the main steam system to the auxiliary steam system (PV-36014A) was placed into service and the auxiliary boilers were secured. At 0930 hours, while at approximately 12% power, operators noticed that auxiliary steam pressure was unstable. At 1015 hours, main feedwater pump (MFP) turbine low pressure steam supply relief valve PSV-30228 started lifting (post-event calibration of PSV-30228 showed that it was within its designed setpoint). Pressure reducing station PV-36014A was placed in manual bypass (manual globe valve MSS-032 was used as the pressure regulating device) in accordance with Casualty Procedure C.22 "Auxiliary Steam Supply Failure." Maintenance personnel were dispatched to repair the valve. The operator assigned to throttle MSS-032 had difficulty communicating with the Control Room due to inaccessibility of a communication station and the high noise level in the area.

At approximately the same time that PV-36014A was placed in manual bypass, pressure control valve PV-30702 became stuck in the partially open position. This valve controls pressure from the auxiliary steam header to the MFPs. The valve normally controls pressure at 90 psig. Operators were dispatched to control pressure manually by throttling the 12" gate valve (ASC-009) directly upstream of PV-30702. No procedural guidance existed for dealing with the failure of PV-30702 by utilizing the upstream gate valve. The operators had difficulty maintaining pressure at 90 psig because the valve was not designed for throttling, and the pressure indicator used by the operator is mounted such that it could not be read from ASC-009. Also, the operators experienced similar difficulties communicating with the Control Room as the operator at MSS-032.

Maintenance personnel examined PV-36014A and determined that the adjustable pilot cap had backed off from its normal position. The problem was corrected and the valve functionally tested in accordance with I.012 "Pneumatic Valve Stroking."

At 1420 hours, operators attempted to return PV-36014A to service by placing the valve into service while simultaneously activating its controller and closing its manual bypass valve (MSS-032). Casualty Procedure C.22 and Operating Procedure A.39 "Auxiliary Steam System" did not address returning PV-36014A to service from a manual bypass configuration. The attempt to return PV-36014A to service, along with the manual throttling of ASC-009, caused fluctuations in the auxiliary steam system that resulted in the loss of steam pressure to the 'B' MFP. This caused the loss of feedwater flow and subsequently caused the Once-Through Steam Generator (OTSG) level to drop to the low level Emergency Feedwater Initiation and Control (EFIC) initiation

setpoint for 'B' auxiliary feedwater (AFW). Having observed the initiation of EFIC, the Control Room operators manually tripped the reactor at 1433 hours in accordance with Casualty Procedure C.10 "Loss of Steam Generator Feed." Operators then performed Emergency Operating Procedures E.01 "Immediate Actions" and E.02 "Vital System Status Verification."

Normal indication of a plant trip (i.e., relief valves lifting) did not occur due to the low power level of the plant. The trip also was not announced over the plant public address system. Consequently, some operations personnel did not recognize immediately that the trip had occurred and their lack of awareness delayed their response to the trip.

Continued pressure fluctuations in the auxiliary steam system caused the speed of the 'B' MFP to increase rapidly and subsequently to trip on high discharge pressure.

The reactor did not stabilize within the post-trip window, but continued to cooldown out of the window because of excessive auxiliary steam loads on the 'B' OTSG and the low decay heat levels. The excessive loads were caused by steam demand from the fourth point heater shell side relief valves. These relief valves lifted due to overpressure on the shell side of the fourth point heaters. The overpressure was caused by loss of feedwater flow to the fourth point heaters.

At 1445 hours, Control Room operators commenced termination of the cooldown by isolating AFW to the 'B' OTSG. This was done in accordance with Emergency Operating Procedure E.05 "Excessive Heat Transfer." After AFW had been isolated, the 'B' OTSG boiled dry as it continued to supply auxiliary steam loads. Allowing the OTSG to boil dry stabilized the reactor temperature. Prior to stabilization, the reactor temperature dropped to of 527 deg. F. The post-trip window temperature limit is 540 deg. F.

At 1450 hours, auxiliary steam to the fourth point heaters was secured. This eliminated the excessive auxiliary steam loads on the 'B' OTSG and allowed the fourth point heater shell side relief valves to close. Also at 1450 hours, the auxiliary boilers were started to maintain the auxiliary steam loads.

At approximately 1500 hours, the operators began to trickle-feed the 'B' OTSG, using AFW, at a rate of approximately 50 gpm. This ended the dry steam generator condition. The OTSG was in a dry condition for approximately 15 minutes. The OTSG water level was approximately 30 inches prior to the event, and reached a peak of 50 inches prior to being isolated. By 1530 hours, the 'B' OTSG pressure returned to 870 psig.

"Red Phone" notification of the event was made to the NRC at 1736 hours on December 12, 1988.

TEXT PAGE 4 OF 8

FIGURE OMITTED - NOT KEYABLE (DRAWING)

Figure 1: AUXILIARY STEAM TO MAIN FEEDWATER TURBINES

The above events are reportable pursuant to 10 CFR 50.73(a)(2)(iv). Automatic initiation of the EFIC System constitutes an automatic actuation of an Engineered Safety Feature (ESF). Manually tripping the reactor is an unplanned manual actuation of the Reactor Protection System.

Cause

The cause of the reactor trip was the dual failure of pressure control valves PV-36014A and PV-30702. The failure of PV-30702 was determined to have resulted from the presence of a foreign object (pin) inside the valve. The pin came from manual globe valve MSS-032 (Refer to Figure 2 for cross sectional view of MSS-032). Post trip radiography of the valve revealed internal damage. Specifically, the valve's pin guide and disc centering pin became detached from the valve. The pin (5/8" O.D. by 4-1/8" long) traveled downstream in the auxiliary steam piping and became lodged inside the body of PV-30702. The pin was caught in the valve in a manner that did not allow the control valve plug to stroke completely closed. This prevented the valve from throttling steam flow at less than approximately 80% open.

TEXT PAGE 5 OF 8

The cause of the damage to the globe valve is a design deficiency stemming from original construction. Neither the selection nor the design of the valve matched the application intended for the valve. The valve was designed to provide manual throttling when its associated pressure reducing station failed. However, to maintain downstream pressure within the desired range for the plant configuration at the time, the valve need only be opened a slight amount. The valve manufacturer recommends using this type of valve for throttling control only at greater than 10 % open. Throttling at an extremely low open position caused excess internal vibration, damage, and eventual failure of the valve.

The failure of PV-36014A was caused by its pilot cap lock nut backing off from the normal position. This valve is located in an area, of high vibration, and the dislocation of its pilot cap lock nut has been. attributed to vibration of the valve.

FIGURE OMITTED - NOT KEYABLE (DRAWING)

Figure 2: 6 Inch Manual Globe Valve

The cause of the reactor cooling to a level outside the post-trip window was the low decay heat levels combined with auxiliary steam loads that are inherent at 12% power. The need to isolate the fourth point heaters manually is a design deficiency which made the plant vulnerable to departing the post-trip window at the 12% level. Post-trip modifications that allow isolation of the fourth point heaters from the Control Room eliminated this deficiency.

The Failure Mode and Consequence of Each Failed Component

Pressure control valve PV-36014A failed in the full open position. Failure of this valve caused pressure fluctuations in the auxiliary steam system and inability to automatically control system pressure downstream of the valve.

Pressure control valve PV-30702 failed in the partially open position. Failure of this valve caused pressure downstream of the valve to be higher than the 90 psig valve control setpoint, and resulted in intermittent lifting of relief valve-PSV-30228 (setpoint 225 psig). However, downstream pressure was well within piping design limits.

Manual globe valve MSS-032 suffered internal damage. The pin that centers the valve disc on its seat broke off. The consequence of this failure was that the pin became lodged in the body of PV-30702, rendering PV-30702 ineffective. The effect this failure may have had on the operator's ability to manually control main steam to auxiliary steam pressure via the globe valve is not known.

A post-trip examination of the oxidation present on the fracture area of the pin indicates that it became detached from the valve prior to December 12, 1988. The plant had been operating for an unknown period of time with the pin and pin guide detached from the valve. Since this valve is not normally used during power operation, this failure had no previous effect. A radiographical search was made downstream of PV-36014A and MSS-032 to locate any other internal valve pieces. None were discovered.

The Energy Industry Identification System (EIIS) Component and System Identifier

The EIIS component function identifier for pressure control valves PV-36014A and PV-30702 is PCV. The EIIS component function identifier for manual globe valve MSS-032 is ISV.

Manufacturer and Model Number

The manufacturer of pressure control valves PV-36014A and PV-30702 is Fisher Controls. The model number for PV-36014A is 657-ED. The model number for PV-30702 is 667-ED. The manufacturer of MSS-032 is Pacific Valves Incorporated. The model number for manual globe valve MSS-032 is 660-U-WE.

Assessment of the Safety Consequences

There was no damage to plant equipment as a result of the three valve failures and subsequent plant trip. The auxiliary steam system was maintained within its design maximum pressure. The Reactor Coolant System was maintained well within its safety limit for excessive heat transfer. The stresses induced on the OTSG due to the dryout were well within its design limit as verified by analysis by the NSSS vendor, Babcock & Wilcox.

There was no impact to the health or safety of the public due to this event.

Corrective Actions

The following major corrective actions have been made as a result of this event:

An assessment was made of the auxiliary steam system pressure downstream of PV-30702. The results of this assessment indicate that the auxiliary steam system downstream of PV-30702 can operate at 200 psig in the event of a fail open condition for PV-30702.

Manual globe valve MSS-032 has been secured closed. Pressure control valve PV-36014A has been modified with a hand wheel for alternate manual positioning and a hand controller. A new three-way air valve (ASC-933) has been added to the auxiliary steam system to enable the failure of the automatic control feature for PV-36014A to be replaced with manual pressure control from PV-36014A.

The adjustable pilot cap on PV-36014A has been lock wired in place to prevent its dislocation in the future.

The two air control regulators associated with PV-36014A have been relocated to an area of reduced vibration levels.

A modification to the fourth point heater auxiliary steam heating isolation valves was performed to allow their isolation to be performed from the Control Room.

Emergency Operating Procedure E.01 was revised to add a step to announce a reactor trip on the public address system.

Emergency Operating Procedure E.02 was revised to:

* Isolate unneeded post-trip steam loads as a post-trip response
* Eliminate possible confusion regarding indications of overcooling
* Add a step to announce a reactor trip on the public address system
* Reduce the probability of drying out an OTSG due to excessive steam loads while rapidly terminating or controlling excessive cooling

Emergency Operating Procedure E.05 was revised to:

* Eliminate possible confusion regarding indications of overcooling
* Prioritize valve isolations during excessive heat transfer
* Reduce the probability of drying out an OTSG due to excessive steam loads while rapidly terminating or controlling excessive cooling

Operations Administrative Procedure OAP-54 was revised to address requirements for communications when a system or components are taken to manual control.

Operating Procedure A.6 was revised to provide improved guidance for trickle feeding OTSGs.

Casualty Procedure C.22 was revised to:

* Reflect the addition of the manual pressure control mode for PV-36014A
* Establish procedural recovery from pressure control valve PV-36014A manual bypass operation
* Address failure of pressure control valve PV-30702
* Provide guidance to consider operating the auxiliary boilers whenever the auxiliary steam pressure reducing station malfunctions

Casualty Procedure C.23 was revised to reflect the addition of the manual pressure control mode for PV-36014A.

Conducted training with each operating crew on:

* Problems that arose during the trip
* Operation of the fourth point heater supply valves

Operations Administrative Procedure OAP-0001 was revised to provide guidance on actions to be taken should the plant be operating in a condition or lineup not addressed by procedures.

A comprehensive program of pre-startup and post-startup corrective actions has been submitted separately to the NRC.

Previous Similar Events

A review of previously submitted LERs indicated that there has been no previous event where failure of the auxiliary steam system led to a plant trip.

ATTACHMENT 1 TO 8901190387 PAGE 1 OF 1

SMUD
SACRAMENTO MUNICIPAL UTILITY DISTRICT 6201 S Street, P.O. Box 15830, Sacramento CA 95852-1830,
(916) 452-3211
AN ELECTRIC SYSTEM SERVING THE HEART OF CALIFORNIA

CEO 89-006

January 11, 1989

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk

Washington, DC 20555

Docket No. 50-312
Rancho Seco Nuclear Generating Station
License No. DPR-54
LICENSEE EVENT REPORT 88-19: MANUAL REACTOR TRIP DUE TO TWO PRESSURE
CONTROL VALVE FAILURES

Attention: George Knighton

In accordance with the requirements of 10 CFR 50.73(a)(2)(iv), the Sacramento Municipal Utility District hereby submits Licensee Event Report 88-19.

Members of your staff who require additional information or clarification may contact Mr. Steven W. Rutter at (209) 333-2935, extension 4911.

Sincerely,

Joseph F. Firlit
Chief Executive Officer
Nuclear

Attachment

cc w/atch: J. B. Martin, NRC, Walnut Creek
A. D'Angelo, NRC, Rancho Seco
INPO

RANCHO SECO NUCLEAR GENERATING STATION
14440 Twin Cities Road, Herald, CA 95638-9799; (209) 333-2935

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